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Effect of Neutron Irradiation on Fatigue Crack Propagation in Types 304 and 316 Stainless Steels at High Temperature

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CONTENTS

ABSTRACT

The resistance of pre- and postirradiation AISI Types 304 and 316 stainless steels to fatigue crack propagation was determined at 800° and 1100°F (427° and 593°C) using the fracture mechanics approach. The effect of irradiation on the fatigue resistance of these steels was dependent upon test temperature and irradiation conditions. In general, irradiation degraded the fatigue resistance at 1100°F (593°C), but at 800°F (427°C) enhancement was also observed. In both steels irradiated in a thermal reactor to a neutron fluence of 1.8×10^{21} n/cm^2 for neutron energies > 0.1 MeV, fatigue crack growth rates at 800°F (427°C) were lower than in the unirradiated steels for a given stress intensity factor range ΔK. However, at 1100°F (593°C) the effect was reversed and crack growth rates were higher in the irradiated steels. Irradiation in a fast reactor to a fluence of $\approx 1.2 \times 10^{22}$ n/cm^2 > 0.1 MeV caused fatigue crack growth rates at 800°F (427°C) to increase at low values of ΔK and decrease at high values of ΔK. At 1100°F (593°C) the crack growth rates in the irradiated steel were either the same as, or higher than, the unirradiated steel. The influence of irradiation on fatigue lives generally reflected the effects observed on crack growth rate.

PROBLEM STATUS

This is a final report on one phase of the problem; work on other phases is continuing.

AUTHORIZATION

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EFFECT OF NEUTRON IRRADIATION ON FATIGUE CRACK PROPAGATION IN TYPES 304 AND 316 STAINLESS STEELS AT HIGH TEMPERATURE

INTRODUCTION

Many structural components of nuclear reactors are subjected to cyclic loads and thus may fail by fatigue. An assessment of structural performance is a critical requirement to assuring fracture safe service of the reactor under operational conditions. An analysis requires that pertinent material properties be evaluated as near as possible to the appropriate service conditions of temperature, stress, irradiation, and coolant environment, as well as to design loads. Although many reactor components are being made of austenitic stainless steels, particularly for fast breeder reactors, yet many aspects of their behavior under service conditions are not known. The fatigue properties of these steels are of immediate interest.

For evaluating the fatigue resistance of materials, the traditional approach has been to relate cycles to failure to the strain or applied stress. In addition, more recently the fracture mechanics approach of relating fatigue crack growth rate to the stress intensity factor has been used. While the traditional approach takes into account both the crack initiation and propagation stages, the fracture mechanics approach involves only crack propagation. Application of the latter approach is based on the presumption that small flaws preexist in the structure and, under cyclic loading, one of the flaws may grow to a large crack and cause failure. Information on crack growth characteristics of the material and applied stresses will permit a calculation of the number of cycles required to cause the flaw to grow to the critical crack size. The fracture mechanics approach, which would tend to give a conservative estimate of life in comparison to the traditional approach, is particularly appropriate for the fatigue analysis of large structures because these structures are likely to contain flaw sources such as welds.

Of the several reactor environmental conditions that might be expected to influence fatigue behavior, the more important are temperature and radiation. A few investigations on the combined effects of these factors on the fatigue life of stainless steels have been reported (1-7), but these have been based on the traditional approach. The findings regarding irradiation effects are varied and depend upon test and irradiation conditions. While several investigations utilizing the fracture mechanics approach (8-11) showed that fatigue crack growth rates are increased by increases in temperature, the effect of radiation on crack growth was not examined.

In this investigation the influence of neutron irradiation on resistance to fatigue crack growth of Types 304 and 316 stainless steels has been determined at elevated temperature. Comparisons have been made between pre- and postirradiation crack growth rates on the basis of the fracture mechanics analysis. Also, the dependence on test temperature of the irradiation effects has been examined.

EXPERIMENTAL PROCEDURE

The chemical compositions of the solution-annealed AISI Types 304 and 316 stainless steel plates used in this study are given in Table 1. Plastic flow properties of the materials at elevated temperatures in the irradiated and unirradiated conditions were

Table 1
Chemical Compositions of Stainless Steels

Steel Type	Chemical Composition (wt-%)										
	C	Mn	P	S	Si	Cr	Ni	Mo	Al	Cu	Other
304	0.048	1.48	0.025	0.015	0.52	18.57	9.45	—	—	—	—
316-A (Ht 65808)	0.060	1.72	0.012	0.007	0.40	17.30	13.30	2.33	0.012	0.065	0.030 Co 0.003 Ti 0.0005 B
316-J	0.07	1.58	0.020	0.007	0.30	17.8	13.4	2.55	0.03	—	0.03 V

determined by means of compression testing; these are listed in Table 2 along with room-temperature (77°F) tensile properties. Of the two Type 316 heats, heat A was irradiated in the Advanced Test Reactor (ATR) and heat J in the Experimental Breeder Reactor-II (EBR-II). Specimens of the Type 304 steel were irradiated in both reactors. Irradiation of the fatigue and compression specimens in the thermal reactor (ATR) was at 550°F (288°C) to a neutron fluence of 1.8×10^{21} n/cm^2 > 0.1 MeV, and in the fast reactor (EBR-II) at the ambient sodium temperature, estimated as 760° ±25°F (404° ±14°C), to a fast fluence of approximately 1.2×10^{22} n/cm^2 > 0.1 MeV.

Single-edge-notch cantilever fatigue specimens containing side grooves, shown in Fig. 1, were used in the investigation. Specimen orientation was such that the plane of crack growth was perpendicular to the rolling direction. For the postirradiation tests, small center sections of approximately 1.2×2.5 in. (30.5×63.5 mm) and 2.2×2.5 in. (55.9×63.5 mm) were irradiated in the ATR and EBR-II, respectively, and then welded in-cell to two end tabs to produce the dimensional requirements of Fig. 1. Many of the unirradiated specimens were prepared in the same way. During welding, temperatures in the region out to 1/4 in. from the groove were maintained below 400°F (204°C). The specimens were cycled at approximately 10 cpm (0.17 Hz) in cantilever bending in a small hydraulic machine having a saw-tooth loading pattern. Loading was varied from zero to tension (R = 0) with the maximum load maintained constant. The tests were conducted in laboratory air.

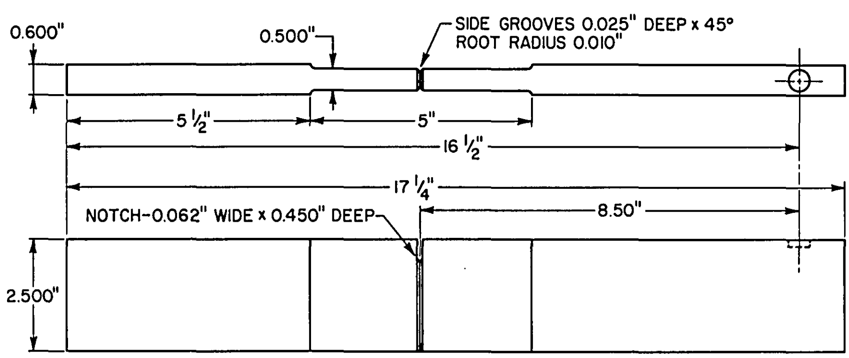

Fig. 1 - Dimensions of single-edge-notch cantilever fatigue specimen

Table 2
Plastic Flow (Compression) Properties

Steel Type	Temperature		Neutron Fluence (10^{21} n/cm^2 > 0.1 MeV)	0.2% Yield Strength		Ultimate Strength		Strain Hardening Coefficient	Uniform Elongation (%)
	(°F)	(°C)		(ksi)	(MN/m^2)	(ksi)	(MN/m^2)		
304	77	25	0	37.0*	255	83.0*	572	—	57†
	800	427	0	24.0	165	64.6	445	0.34	40
			1.8	74.0	510	81.7	563	0.14	15
	1100	593	0	23.9	165	52.8	364	0.29	34
			1.8	44.5	307	58.7	404	0.14	15
			≈ 12.	86.8	598	88.9	613	0.05	5
316-A	77	25	0	44.1*	304	82.1*	566	—	68†
	800	427	0	28.6	197	65.1	449	0.34	40
			1.8	78.8	543	82.8	570	0.09	9
	1100	593	0	23.7	163	61.1	421	0.32	38
			1.8	44.5	307	63.9	440	0.23	26
316-J	800	427	0	24.7	170	59.2	408	0.34	40
			≈ 12.	72.7	501	86.3	595	0.14	15

*Tensile data.
†Tensile total elongation.

Heating of the specimen was accomplished by means of a 5-in.-long induction coil placed around the grooved test section. An opening between the coil turns permitted observation of the crack for length measurements. Specimen temperature was monitored by spot-welded thermocouples which showed that the gradient along the side groove was usually no greater than 10°F (6°C) and the gradient normal to the groove was no greater than 30°F (17°C) measured 1 in. (25.4 mm) out from the center section. During the test, temperature variations were maintained to within ±5°F (±3°C). Because an automatic crack measurement device was not used, tests were frequently interrupted during the night and then restarted in the morning after holding at temperature for 10 to 15 min. No noticeable effect on the rate of crack growth owing to these interruptions was evident.

During the tests the crack length was measured at the root surface of a side groove by means of a measuring microscope for the out-of-cell tests, and by means of a remote high-resolution television system for the in-cell tests. The television camera followed the tip of the growing crack by lateral displacement, which was automatically measured. Distances of less than 0.001 in. (0.025 mm) could be resolved by the television system. Visibility of the crack was aided by electropolishing of the groove surface and then lightly sand blasting. Measurements of crack length were made until the length reached about 1.5 in. (38.2 mm), including the notch, or until gross deformation was observed in the vicinity of the crack tip, at which point the test was terminated. The rates of crack growth were determined from measuring the slopes of the plotted curve of crack length versus number of cycles.

Analysis of the fatigue crack growth data was based on the power law proposed by Paris and Erdogan (12). This law relates crack growth rate da/dN to the range of stress intensity factor ΔK in the generalized form

$$da/dN = C(\Delta K)^m$$

where C and m are constants dependent upon material, temperature, and load conditions. That this expression describes fatigue crack growth data of metals at room and elevated temperatures at least over limited ranges of ΔK has been well demonstrated. Stress intensity values for this study were computed using the formula for pure bending given by Gross and Srawley (13) with a correction for the face grooves. The formula is

$$K = \frac{6PL}{(B\times B_N)^{1/2}\, W^{3/2}}\, Y$$

where

$$Y = 1.99(a/W)^{1/2} - 2.47(a/W)^{3/2} + 12.97(a/W)^{5/2} - 23.17(a/W)^{7/2} + 24.80(a/W)^{9/2},$$

and P is the maximum cyclic load, L is the distance from the plane of the crack to the point of load application, a is the total length of crack and notch, W is the specimen width, B is the specimen thickness, and B_N is the net thickness at the grooves. The computed values agree with those given by Kies' equation (Ref. 14) to within 3.3% over the a/W range of application. For the computations, corrections were not made for plasticity at the crack tip.

RESULTS

Thermal Reactor Environment

For analysis of the fatigue crack growth behavior, the crack growth rates da/dN of the irradiated and unirradiated stainless steels at 800° and 1100°F (427° and 593°C) were plotted as a function of stress intensity factor range ΔK on log-log coordinates (Figs. 2-5). On these plots the data fit straight lines with slope transitions at both low and high values of ΔK and crack growth rates. The power-law relationships for the irradiated and unirradiated steels are given for the intermediate growth rate range of approximately 10^{-5} to 10^{-4} in./cycle (3×10^{-4} to 3×10^{-3} mm/cycle). Generally, a wider range of growth rates is covered by the given relationship for the irradiated steel. That is, the irradiation appears to have extended the range at intermediate growth rates, which can be described by a single power law.

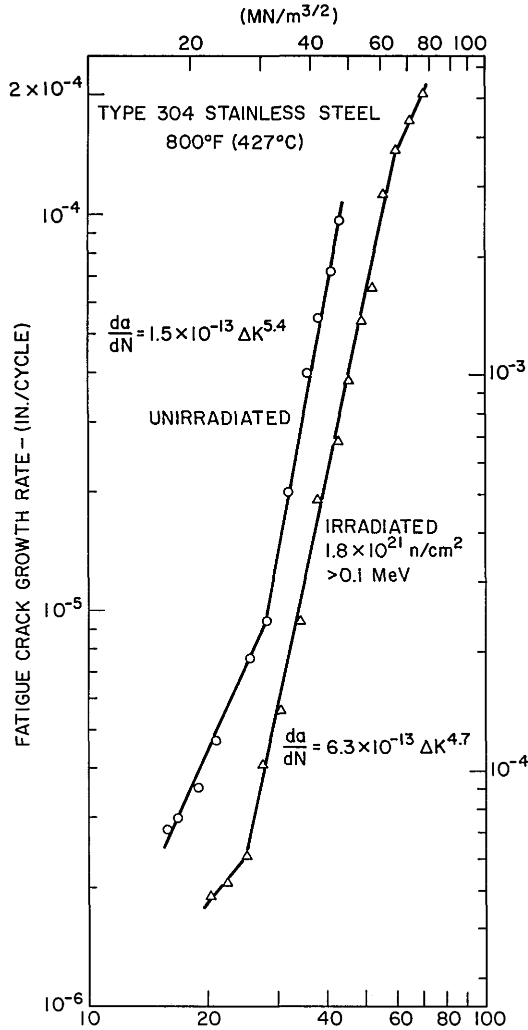

Fig. 2 - Influence of irradiation (1.8×10^{21} n/cm² > 0.1 MeV in ATR) on fatigue crack growth rates for Type 304 stainless steel at 800°F (427°C)

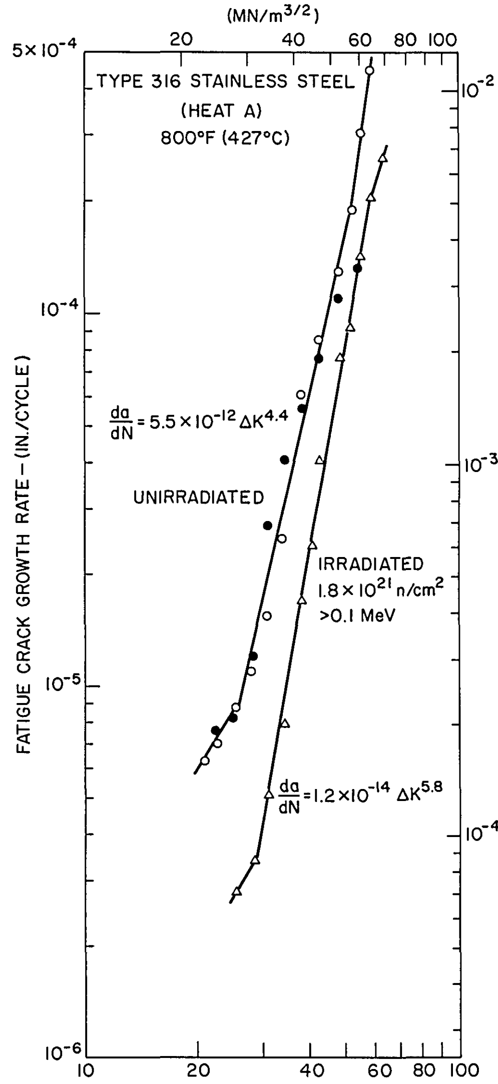

Fig. 3 - Influence of irradiation (1.8×10^{21} n/cm² > 0.1 MeV in ATR) on fatigue crack growth rates for Type 316 stainless steel at 800°F (427°C)

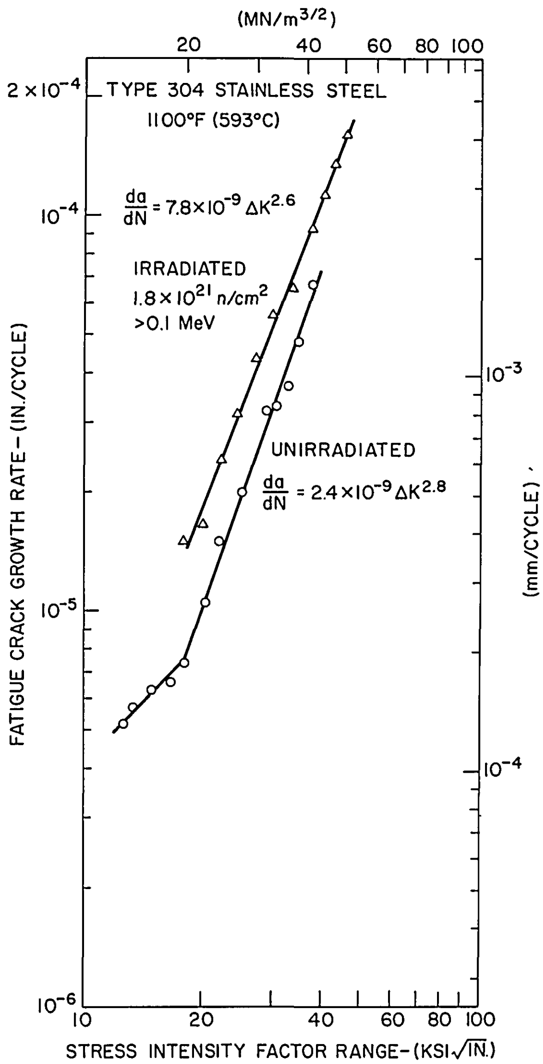

Fig. 4 - Influence of irradiation (1.8×10^{21} n/cm^2 > 0.1 MeV in ATR) on fatigue crack growth rates for Type 304 stainless steel at 1100°F (593°C)

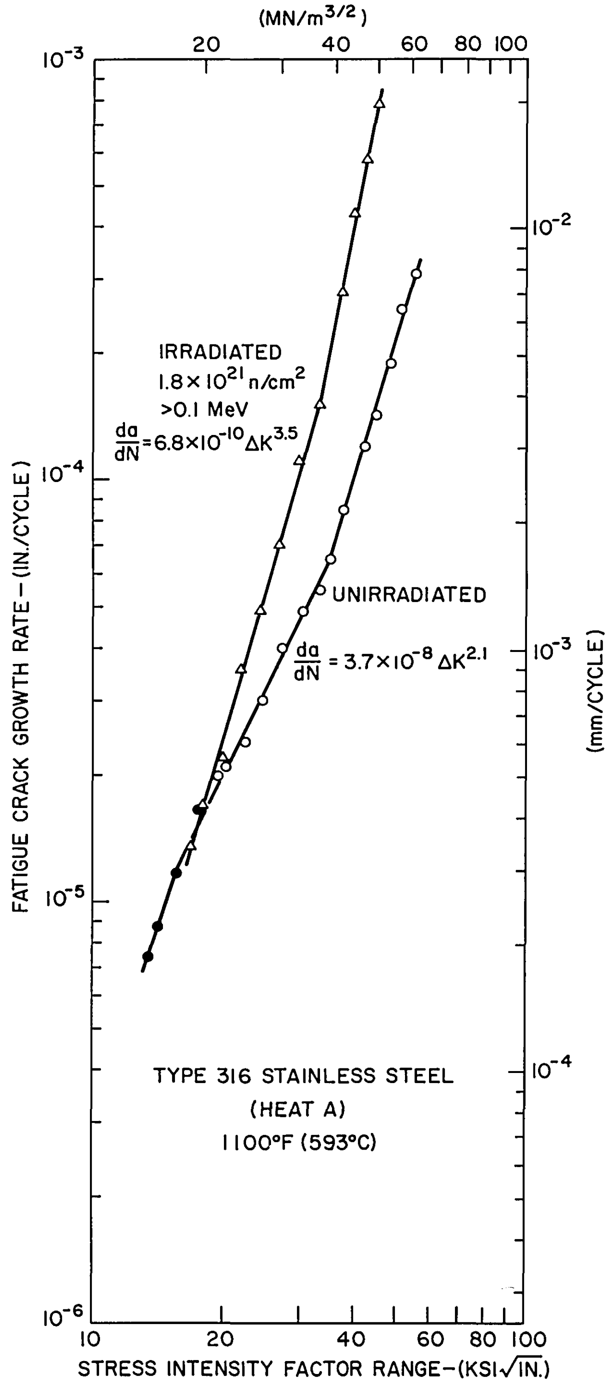

Fig. 5 - Influence of irradiation (1.8×10^{21} n/cm^2 > 0.1 MeV in ATR) on fatigue crack growth rates for Type 316 stainless steel at 1100°F (593°C)

The effect of the thermal reactor radiation was to reduce crack growth rate for a given ΔK at 800°F (427°C), as shown in Figs. 2 and 3. Both Types 304 and 316 stainless steels responded similarly, the growth rates in the irradiated steel being about 1/4 to 1/2 of that in the unirradiated steel. The increase in da/dN with an increase in ΔK, shown by the exponent m, was not significantly influenced by irradiation over the intermediate growth rate range. A slight decrease in m owing to irradiation was indicated for Type 304 steel compared with a slight increase for the Type 316 steel. Actually, Types 304 and 316 steels in the irradiated condition had very nearly equivalent crack growth rates for a given ΔK at 800°F (427°C), and this corresponds to their comparable preirradiation resistances to crack growth.

With an increase in test temperature from 800° to 1100°F (427° to 593°C) the crack growth rates da/dN for both steels increased. The magnitude of the increase was considerably greater for the irradiated steel than for the unirradiated steel (Figs. 4 and 5). For example, the growth rates in irradiated Type 304 steel were raised by as much as a factor of 10 because of the temperature increase. At 1100°F (593°C) the effect of irradiation was the converse of that at 800°F (427°C) and crack growth rates were usually higher in the irradiated steels. The power-law exponents m of the irradiated and unirradiated materials were alike for the Type 304 steel, but differed for the Type 316 steel. Yet the exponents for all of the materials were lower for the 1100°F (593°C) than the 800°F (427°C) data. At 1100°F (593°C) the crack growth rates in Type 316 steel were higher than in Type 304 steel in the same condition, in contrast to their comparable resistances at 800°F (427°C).

In general, neutron irradiation in a thermal reactor operating at 550°F (288°C) improved the resistance to fatigue crack growth at 800°F (427°C) but degraded the resistance at 1100°F (593°C).

With regard to the plastic flow properties, the irradiation raised the yield strength and lowered the ductility of the two stainless steels similarly, as shown in Table 2. Yield strengths of the irradiated steels were considerably higher at 800°F (427°C) than at 1100°F (593°C). Removal of some displacement damage at 1100°F (593°C) probably accounts for the much smaller increase in yield strength at the higher temperature.

Fast Reactor Environment

Irradiation in the EBR-II (fast neutron fluence $\approx 1.2 \times 10^{22}$ n/cm^2 > 0.1 MeV) altered the fatigue crack growth characteristics of Type 316 stainless steel at 800°F (427°C), as shown in Fig. 6. Crack growth rates were higher in the irradiated steel than in the unirradiated steel at values of $\Delta K < 40$ ksi $\sqrt{\text{in.}}$ (43.9 MN/m$^{3/2}$) and lower at $\Delta K > 40$ ksi $\sqrt{\text{in.}}$ In effect, the power-law exponent m was significantly lower for the irradiated steel — 2.9 compared to 5.1 for the unirradiated condition. A single power relationship covers a wide range of data.

At 1100°F (593°C) the power-law exponent was also lower for the irradiated than unirradiated Type 304 steel (Fig. 7), but the difference was smaller than for Type 316 steel at 800°F (427°C). Crack growth rates were higher in the irradiated material at low ΔK values, though they were the same as in the unirradiated material at high ΔK values. Relatively small effects on growth rates at low ΔK values, and consequently low rates, may be reflected as appreciable effects on fatigue life. This results from the fact that the growth of small cracks at low rates comprises a relatively large share of the cycles to failure. In fact, the fatigue lives* of the Type 304 steel at 1100°F (593°C) and

*Fatigue life is taken as the number of cycles required to increase actual crack length to 1.1 in.

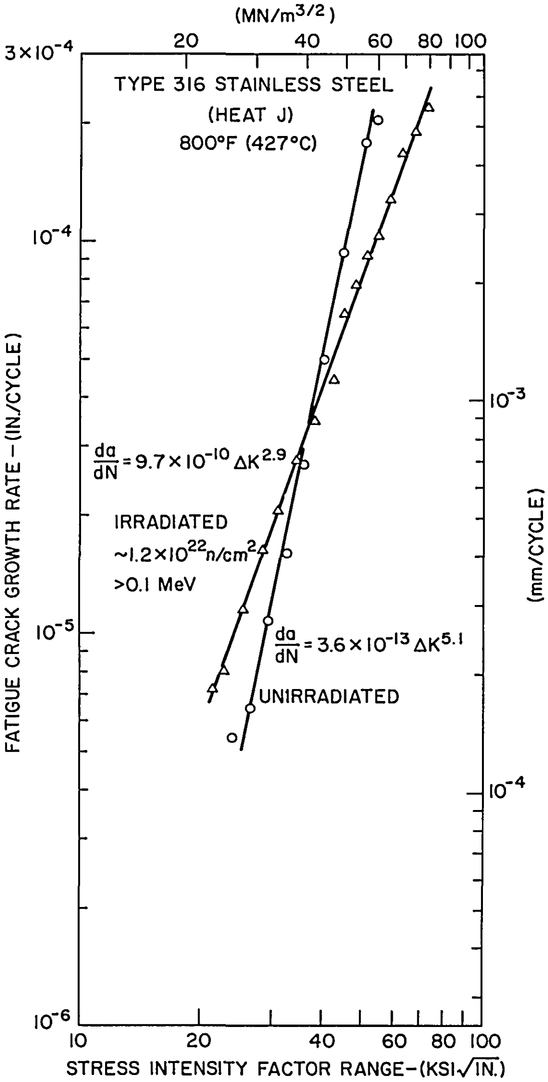

Fig. 6 - Effect of irradiation ($\approx 1.2 \times 10^{22}$ n/cm^2 > 0.1 MeV in EBR-II) on fatigue crack growth rates for Type 316 stainless steel at 800°F (427°C)

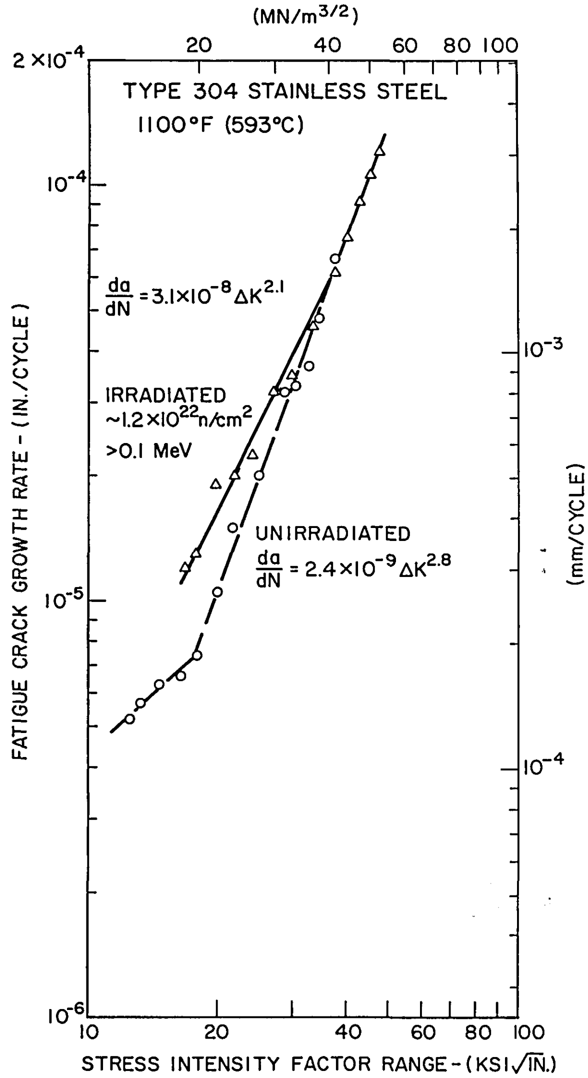

Fig. 7 - Effect of irradiation ($\approx 1.2 \times 10^{22}$ n/cm^2 > 0.1 MeV in EBR-II) on fatigue crack growth rates for Type 304 stainless steel at 1100°F (593°C)

700 lb (3115 N) in the irradiated and unirradiated conditions (Table 3) were 56 and 36 kcycles, respectively. Further evidence of this effect may be seen in Fig. 8 in the crack growth curves which show that fatigue life was shorter for the irradiated than the unirradiated Type 316 steel. It appears that in the Type 316 steel at 800°F (427°C), shown in Fig. 8, only 20% of the crack extension occurred in the first 50% of the total cycles. On examination of the corresponding crack growth rate data, shown in Fig. 6, the effect on fatigue life is not readily evident. Of course, the irradiation effect on life will depend upon the initial flaw size and load.

Table 3
Irradiation Effect on Fatigue Life

Steel Type	Temperature		Load		Neutron Fluence (10^{21} n/cm^2 > 0.1 MeV)	Life (kcycles)
	(°F)	(°C)	(lb)	(N)		
304	800	427	650	2893	0	174
			800	3560	1.8 (ATR)	203
	1100	593	700	3115	0	56
					1.8 (ATR)	29
					≈ 12. (EBR-II)	36
316-A	800	427	900	4005	0	56
					1.8 (ATR)	139
	1100	593	700	3115	0	23
					1.8 (ATR)	21
316-J	800	427	900	4005	0	59
					≈ 12. (EBR-II)	47

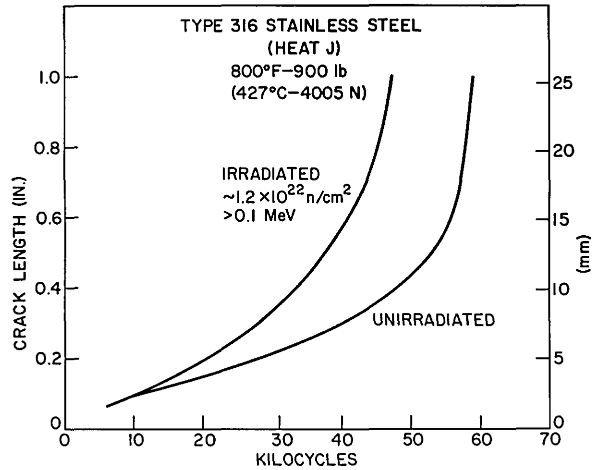

Fig. 8 - The fatigue crack growth curves for Type 316 stainless steel at 800°F (427°C) show faster growth and shorter life for the irradiated (EBR-II) than the unirradiated condition

With respect to effects on flow properties, the yield strength was raised and the ductility was reduced by the irradiation. For these EBR-II irradiated steels, the strength increase was at least as large at 1100°F (593°C) as at 800°F (427°C), in contrast to the substantial strength drop at the higher test temperature for the ATR irradiated steel.

Comparison Between Thermal and Fast Reactor Irradiations

Examination of the relative effects of irradiation in the thermal reactor (ATR) and the fast reactor (EBR-II) shows that the effects depend upon test temperature. At 800°F (427°C) the fatigue life of Type 316 steel is increased considerably after the ATR irradiation, but is reduced slightly after the higher temperature higher fluence EBR-II irradiation (Table 3). Also, crack growth rates generally were correspondingly lower in the ATR irradiated material (Figs. 3 and 6).

A comparison between the effects of irradiation on fatigue crack growth and fatigue life at 1100°F (593°C) in the ATR and EBR-II is shown in Fig. 9 for Type 304 steel. While fatigue life was reduced by both irradiations, the reduction was larger for the lower temperature lower fluence ATR irradiation. In agreement with this result, the crack growth rates for a given ΔK were higher for the ATR irradiation, as may be seen from a comparison of Figs. 4 and 7.

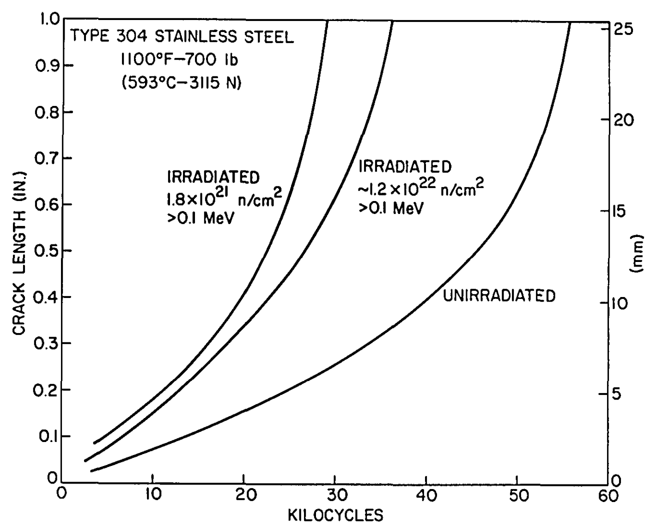

Fig. 9 - Comparison of the effects of irradiating Type 304 stainless steel in the ATR and the EBR-II on fatigue crack growth at 1100°F (593°C)

DISCUSSION

The effect of thermal reactor irradiation in causing the reversal from enhanced fatigue resistance at 800°F to a degradation at 1100°F suggests that two separate processes are involved. One of these processes is probably dominant at the lower temperature, while at the higher temperature the other process is more effective. Neutron-irradiation-produced property changes are usually attributed to either displacement damage or to helium generated from transmutation reactions. Both of these may have a part in the observed irradiation effects on fatigue behavior.

The observed improvement in resistance to fatigue crack growth due to irradiation of the stainless steels at 800°F (427°C) most likely results from displacement damage. Displacement damage usually raises the yield strength and lowers the ductility, both of which would tend to reduce the size of the plastic zone at the crack tip and, in turn, reduce crack growth rate. Smaller plastic zones in an alloy steel have been correlated with lower crack growth rates (15).

With an increase in temperature to 1100°F (593°C), displacement damage produced at 550°F (288°C) is largely (but not completely) annealed out, as indicated by the decrease in yield strengths in Table 2. However, helium remains at the higher temperature and may cause the decrease in crack growth resistance. Helium embrittlement of grain boundaries is a possible mechanism by which crack propagation may be facilitated, as an easier crack path is provided. The effects of helium on other properties are well documented. For example, ductility losses of irradiated stainless steel at high temperatures have been ascribed primarily to the presence of helium, usually from agglomeration at grain boundaries (16-19). Also, reductions in creep-rupture life of irradiated stainless steel have been attributed at least in part to the fact that helium promotes boundary cracking (20,21). In fact, cyclotron-injected helium has been shown (22) to reduce creep-rupture life and rupture strain of Type 304 steel at 1112°F (600°C). Implanted helium has been found (23) also to reduce the fatigue life of Type 304 steel at 1500°F (816°C), but grain boundary embrittlement was not evident because the fracture was transgranular. However, Beeston and Brinkman (5) reported that fatigue cracking in neutron irradiated Type 316 steel at 1112°F (600°C) included both transgranular and intergranular modes as the material experienced a reduction in fatigue life. Although the effect of helium occurs mainly at temperatures above 1100°F (593°C) and through grain boundary embrittlement, these data indicate that helium can affect properties at 1100°F (593°C), the temperature of interest in this study, and the effect need not be related to boundary cracking.

Another possible explanation of our observed irradiation-enhanced fatigue crack growth at 1100°F (593°C), in addition to helium embrittlement, is that irradiation-produced defects, including helium, within grains influence the deformation characteristics of the plastic zone at the crack tip and, in turn, the crack propagation. Fatigue-induced substructure changes as a result of irradiation have been noted (5).

The data of this study are generally consistent with those reported by other investigators for irradiation effects on the fatigue life of stainless steels. It appears that, in general, fatigue life — on the basis of total strain — is improved at low temperature by irradiation at fluences in the range of 10^{21} n/cm^2 > 0.1 MeV, or less, and is degraded at high temperature. Increases in fatigue life due to neutron irradiation have been observed in tests at room temperature (1,2) and at 550°F (288°C) (Ref. 7). On the other hand, decreases in fatigue life have usually been noted at temperatures of 932°F (500°C) and higher (3-6). Our findings of enhanced fatigue resistance at 800°F (427°C) and degraded resistance at 1100°F (593°C) by thermal reactor irradiation are in agreement. However, the slight decrease in fatigue resistance at 800°F (427°C) by fast reactor radiation indicates that, with higher fluences, the degradation of fatigue resistance can occur at lower temperatures.

CONCLUSIONS

The study of irradiation effects on fatigue resistance of austenitic stainless steel yielded the following conclusions:

1. Fatigue crack growth rates are exponentially related to the stress intensity factor range ΔK for the irradiated, as well as unirradiated, stainless steels. Irradiation extends the intermediate crack growth rate range over which a single power law is applicable.

2. Irradiation in a thermal reactor to a neutron fluence of 1.8×10^{21} n/cm^2 > 0.1 MeV improved the resistance to crack growth at 800°F (427°C) for Types 304 and 316 stainless steels. The two steels in the irradiated condition had approximately equivalent crack growth rates.

3. At 1100°F (593°C), however, the resistance to crack growth was generally better for the unirradiated steels than for the thermal reactor irradiated steels.

4. Crack growth rates were more sensitive to temperature in the range of 800° to 1100°F (427° to 593°C) after irradiation of the steels.

5. Neutron irradiation in a fast reactor to a fluence of $\approx 1.2\times10^{22}$ n/cm^2 > 0.1 MeV caused an increase in crack growth rates at low values of ΔK at 800° and 1100°F (427° and 593°C). However, at high ΔK values, growth rates were lower in the irradiated steel at 800°F (427°C), and about the same at 1100°F (593°C), for the irradiated and unirradiated conditions.

6. Fatigue lives generally reflected changes in crack growth rates, with the irradiation effects at the low ΔK range being more important. At 1100°F (593°C) the fatigue lives of the irradiated steels were shorter than those of the unirradiated steels.

ACKNOWLEDGMENTS

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REFERENCES

1. Kangilaski, M., and Shober, F.R., "The Effect of Neutron Irradiation on the Mechanical Properties of AISI Type 347 Stainless Steel," ASTM STP 426, pp. 487-511, Philadelphia: Am. Soc. for Testing and Materials, 1967

2. Waldman, L.A., and Doumas, M., "Fatigue and Burst Tests on Irradiated In-Pile Stainless-Steel Pressure Tubes, Nuclear Appl., 1: 439-452 (Oct. 1965)

3. Reynolds, M.B., "Strain-Cycle Phenomena in Thin-Wall Tubing," ASTM STP 380, pp. 323-336, Philadelphia: Am. Soc. for Testing and Materials, 1965

4. Conway, J.B., Berling, J.T., and Stenz, R.H., "New Correlations Involving the Low-Cycle Fatigue and Short-Term Tensile Behavior of Irradiated and Unirradiated 304 and 316 Stainless Steel," GEMP-726, General Electric Co., Cincinnati, Ohio, Dec. 1969

5. Beeston, J.M., and Brinkman, C.R., "Axial Fatigue of Irradiated Stainless Steels Tested at Elevated Temperatures," ASTM STP 484, pp. 419-450, Philadelphia: Am. Soc. for Testing and Materials, 1971

6. Moteff, J., Smith, J.P., and Stapp, W.J., "Effects of Neutron Irradiation on the Plastic Fatigue Properties of Stainless Steel," Trans. Am. Nuclear Soc., 12(No. 2): 585 (1969)

7. Smith, H.H., and Shahinian, P., "Fatigue Behavior of Irradiated Thin-Section Type 348 Stainless Steel at 550°F (288°C)," presented at ASTM Symposium on Effects of Radiation on Structural Materials, Los Angeles, Calif., June 1972

8. James, L.A., and Schwenk, E.B., Jr., "Fatigue-Crack Propagation Behavior of Type 304 Stainless Steel at Elevated Temperatures," Met. Trans. 2: 491-496 (1971)

9. Shahinian, P., Smith, H.H., and Watson, H.E., "Fatigue Crack Growth in Type 316 Stainless Steel at High Temperature," J. Engineering for Ind., ASME, 93(Ser. B, No. 4):976-980 (1971)

10. Shahinian, P., Watson, H.E., and Smith, H.H., "Fatigue Crack Growth in Selected Alloys for Reactor Applications," J. Materials (to be published)

11. James, L.A., "The Effect of Elevated Temperature Upon the Fatigue-Crack Propagation Behavior of Two Austenitic Stainless Steels," presented at International Conference on Mechanical Behavior of Materials, Kyoto, Japan, Aug. 1971

12. Paris, P., and Erdogan, F., "A Critical Analysis of Crack Propagation Laws," J. Basic Engineering, ASME, 85(Ser. D, No. 4):528-534 (1963)

13. Gross, B., and Srawley, J.E., "Stress-Intensity Factors for Single-Edge-Notch Specimens in Bending or Combined Bending and Tension by Boundary Collocation of a Stress Function," NASA-TN-D2603, National Aeronautics and Space Administration, Washington, D.C., Jan. 1965

14. Kies, J.A., Smith, H.L., Romine, H.E., and Bernstein, H., "Fracture Testing of Weldments," ASTM STP 381, pp. 328-356, Philadelphia: Am. Soc. for Testing and Materials, 1965

15. McHenry, H.I., "Fatigue Crack Propagation in Steel Alloys at Elevated Temperature," ERR-FW-1029, General Dynamics Corp., Sept. 1970

16. Barnes, R.S., "Mechanisms of Radiation-Induced Mechanical Property Changes," ASTM STP 380, pp. 40-67, Philadelphia: Am. Soc. for Testing and Materials, 1965

17. Ward, A.L., and Holmes, J.J., "Ductility Loss in Fast Reactor Irradiated Stainless Steel," Nuclear Appl. and Tech., 9:771-772 (Nov. 1970)

18. Kangilaski, M., Spretnak, J.W., Bauer, A.A., and Wullaert, R.A., "Influence of Irradiation Temperature on the Tensile Properties of Stainless Steel," ASTM STP 484, pp. 194-214, Philadelphia: Am. Soc. for Testing and Materials, 1971

19. Martin, W.R., and Weir, J.R., "Solutions to the Problems of High Temperature Irradiation Embrittlement," ASTM STP 426, pp. 440-456, Philadelphia: Am. Soc. for Testing and Materials, 1967

20. Bloom, E.E., and Stiegler, J.O., "Effect of Fast Neutron Irradiation on the Creep Rupture Properties of Type 304 Stainless Steel at 600°C," ASTM STP 484, pp. 451-466, Philadelphia: Am. Soc. for Testing and Materials, 1971

21. Lovell, A.J., and Barker, R.W., "Uniaxial and Biaxial Creep Rupture of Type 316 Stainless Steel After Fast Reactor Irradiation," ASTM STP 484, pp. 468-483, Philadelphia: Am. Soc. for Testing and Materials, 1971

22. King, R.T., "Cyclotron Simulation of Neutron-Transmutation Produced Gases in Reactor Cladding and Structural Materials," pp. 294-315 in Proceedings, International Conference on the Use of Cyclotrons in Chemistry, Metallurgy, and Biology, Oxford, England, Sept. 1969

23. Serpan, C.Z., Jr., and Smith, H.H., "Fatigue Properties of Stainless Steel after Cyclotron Injection of Helium," pp. 17-19, Report of NRL Progress, Naval Research Laboratory, Washington, D.C., Dec. 1971

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4. DESCRIPTIVE NOTES *(Type of report and inclusive dates)*

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13. ABSTRACT

The resistance of pre- and postirradiated AISI Types 304 and 316 stainless steels to fatigue crack propagation was determined at 800° and 1100°F (427° and 593°C) using the fracture mechanics approach. The effect of irradiation on the fatigue resistance of these steels was dependent upon test temperature and irradiation conditions. In general, irradiation degraded the fatigue resistance at 1100°F (593°C), but at 800°F (427°C) enhancement was also observed. In both steels irradiated in a thermal reactor to a neutron fluence of 1.8×10^{21} n/cm^2 for neutron energies >0.1 MeV, fatigue crack growth rates at 800°F (427°C) were lower than in the unirradiated steels for a given stress intensity factor range ΔK. However, at 1100°F (593°C) the effect was reversed and crack growth rates were higher in the irradiated steels. Irradiation in a fast reactor to a fluence of $\approx 1.2 \times 10^{22}$ n/cm^2 >0.1 MeV caused fatigue crack growth rates at 800°F (427°C) to increase at low values of ΔK and decrease at high values of ΔK. At 1100°F (593°C) the crack growth rates in the irradiated steel were either the same as, or higher than, the unirradiated steel. The influence of irradiation on fatigue lives generally reflected the effects observed on crack growth rate.

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